NRC's Use of Level 3 PRA Information in Severe Accident Mitigation Alternatives Reviews

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Abstract

In the United States, severe accident mitigation alternatives (SAMA) evaluations are prepared by the nuclear industry and reviewed by the NRC staff for three activities: during the license renewal process for existing reactors; during the design certification (DC) application process for new reactors; and during the construction/operating licensing (COL) application process for new reactor licenses. A SAMA is a feature or action that would prevent or mitigate the consequences of a severe accident, and includes potential hardware modifications called severe accident mitigation design alternatives (SAMDA), procedure changes, and training program improvements. Cost-benefit assessments are done to determine if a SAMA should be implemented. Using results from the applicants' Level 2 PRAs as inputs to the MACCS2 code to determine offsite consequences, the applicants are then able to estimate the maximum attainable benefit (MAB) from the SAMAs and determine which, if any, are cost-beneficial. Details on how the MAB is calculated are presented below, using a new plant example. The NRC staff reviews the analyses and prepares safety evaluation reports. Insights from the NRC staff's reviews of some of the SAMA evaluations are presented below, as are some results from an actual new reactor SAMA evaluation.

1. INTRODUCTION

In 1980 NRC issued an interim policy statement on the consideration of severe accidents in environmental impact statements (EISs) applicable to Construction Permit and Operating License applications submitted on or after July 1, 1980 [1]. The policy statement states that it is "the intent of the Commission that the staff takes steps to identify additional cases that might warrant early consideration of either additional features or other actions which would prevent or mitigate the consequences of severe accidents." These features have become known as severe accident mitigation design alternatives (SAMDAs) when applied at the design stage, or SAMAs when applied in the context of extending an existing license.

In August 1985, NRC issued its policy statement on severe reactor accidents that presented NRC's conclusions that existing plants pose no undue risk to public health and safety and that there was no present basis for immediate action on generic rulemaking or other regulatory changes for those plants because of severe accident risk. However, it required each licensee to perform an analysis to discover instances of particular vulnerability to core melt or unusually poor containment performance given a core-melt accident. This was considered to be a sufficient basis for not requiring SAMDAs at the operating license review stage for previously constructed plants. However, a 1989 court decision ruled that consideration of SAMDAs is required for plant operation [2].

Relative to the evaluation of potential improvements for existing reactors in the U.S., the NRC gained considerable experience during the 1980s and 1990s via (a) staff assessments of SAMDAs for the Limerick, Comanche Peak, and Watts Bar plants performed as a result of the aforementioned *Limerick*

Ecology Action court decision, (b) the containment performance improvement program[1], (c) the individual plant examination (IPE) program[2], and (d) the implementation of severe accident management programs at all nuclear power plants as part of an industry initiative. These regulatory programs and initiatives provide assurance that any major vulnerabilities to severe accidents have been identified and addressed, and that the residual level of risk is low. As a result, major plant modifications would not be expected as a result of a SAMA analysis.

All applications for license renewal must consider SAMAs – more than 50 have been completed to date and more are still under development. For new reactor designs, the NRC requires that all applications evaluate and possibly include severe accident prevention and mitigation design features. Part of this requirement includes evaluating SAMDAs and documenting the evaluations in environmental reports. SAMDAs have been completed for all of the designs submitted to the NRC for certification. In many cases contentions from the public have been admitted into the licensing proceedings.

2. MAJOR STEPS IN A SAMA EVALUATION

2.1 Identification and characterization of leading contributors to risk

The first step is to identify and characterize the leading contributors to core damage frequency (CDF) and offsite risk. Maximum use is made of the plant-specific risk model for characterizing the dominant contributors to risk and identifying candidate SAMAs to address these contributors. A simplified approach is generally used to account for external events and analysis uncertainties, although for some of the design certification SAMDAs, the CDFs for external events and shutdown events are included. Benefits are typically quantified using the internal events at power model and then multiplied by the ratio of total CDF to internal event CDF (typically a factor of about 2 but could be as high as 10) to account for external events benefits. In some cases, the SAMA may specifically relate only to external events (e.g., a modification related to a piece of hardware that is only damaged during seismic events). In other cases, a SAMA that may have been identified based on internal event considerations (e.g., use of portable generators to power equipment in a station blackout (SBO)) may also have benefits in externally initiated events (e.g., a seismic induced SBO).

Release categories are defined in the Level 2 PRA, and the associated source terms are computed from accident progression analyses using a code such as MAAP4 or MELCOR. The MACCS2 code is used to determine off-site consequences for each of the release categories. For example (Comanche Peak COL application to build and operate a US-APWR), the CDF for internal events at power is 1.2E-06/ry. The release categories from the US-APWR Level 2 PRA are shown in Table 1.

[1] NRC examined each of five U.S. reactor containment types (BWR Mark I, II and III; PWR Ice Condenser; and PWR Dry) with the purpose of examining the potential failure modes, potential enhancements, and the cost benefit of such enhancements. This examination has been called the containment performance improvement (CPI) program and was documented in a series of reports (NUREG/CR-5225; NUREG/CR-5278; NUREG/CR-5528; NUREG/ CR-5529; NUREG/CR-5565; NUREG/CR-5567; NUREG/CR-5575; NUREG/CR-5586; NUREG/CR-5589; NUREG/CR-5602; NUREG/CR-5623; NUREG/ CR-5630).

[2] In accordance with NRC's policy statement on severe accidents, each U.S. licensee was requested to perform an individual plant examination (IPE) to look for vulnerabilities to both internal and external initiating events (Generic Letter 88-20, Supplements 1-4). These examinations consider potential improvements on a plant-specific basis. Results are described in NUREG-1560 and NUREG-1742, respectively.

Table 1. US-APWR Release Categories for Internal Events at Power

Designator	Description	Release Frequency (per reactor-year)
RC1	Containment Bypass Includes SGTR initiating events and induced SGTR.	7.5E-09
RC2	Containment Isolation Failure	2.1E-09
RC3	Containment Failure Before Core Damage Overpressure due to loss of containment heat removal.	2.0E-08
RC4	Early Containment Failure Due to dynamic loads including early hydrogen combustion, steam explosions, and DCH.	1.1E-08
RC5	Late Containment Failure Includes late overpressure, hydrogen combustion, and basemat melt-through	6.5E-08
RC6	Intact Containment No containment failure. Releases at design leak rate.	1.1E-06
Total		1.2E-06

Table 2 shows the release fractions for the various release categories as computed by MAAP4, for input into MACCS2. There are two plumes for each release category.

Table 2. Source Term Release Fractions

Release Category[b]	Plume No.	Kr/Xe	I	Cs	Te/Sb	Sr	Ru	La	Ce	Ba
RC1[c],[d]	1	9.4E-1	2.8E-1	2.0E-1	1.3E-1	4.9E-3	1.8E-2	2.4E-4	2.8E-4	1.2E-2
RC1[e]	2	7.6E-3	6.3E-3	1.1E-2	8.5E-3	3.9E-3	4.3E-3	2.7E-3	1.9E-3	3.6E-3
RC2[f]	1	9.7E-1	6.8E-2	2.6E-2	4.3E-2	5.4E-3	1.6E-2	4.0E-3	2.3E-3	8.6E-3
RC2	2	2.7E-2	2.1E-1	1.7E-2	3.5E-2	2.3E-3	1.0E-4	1.1E-4	4.1E-4	2.6E-3
RC3[g]	1	9.9E-1	4.8E-1	4.7E-1	4.3E-1	4.4E-2	2.8E-1	1.6E-3	6.4E-3	1.1E-1
RC3	2	2.0E-3	1.3E-3	1.1E-3	4.3E-3	4.9E-4	1.8E-4	6.6E-6	6.3E-5	2.5E-4
RC4[h]	1	1.0E+0	5.5E-2	4.2E-2	5.3E-2	4.8E-3	2.7E-2	1.2E-4	3.7E-4	2.4E-2
RC4	2	3.8E-4	1.4E-2	4.5E-3	1.1E-2	1.3E-3	1.1E-5	1.5E-5	4.7E-4	4.7E-4
RC5[i]	1	9.6E-1	2.5E-2	5.3E-3	9.0E-3	8.2E-5	1.0E-4	3.0E-5	1.9E-5	6.8E-5
RC5	2	2.5E-2	1.2E-1	1.5E-2	7.7E-3	2.2E-6	2.6E-6	5.9E-8	5.9E-8	5.0E-6
RC6[j]	1	7.8E-4	1.7E-6	1.7E-6	1.3E-6	1.7E-7	6.4E-7	3.5E-9	5.6E-9	2.7E-7
RC6	2	1.3E-3	1.9E-9	0.0E+0	6.0E-10	6.5E-11	4.4E-11	4.6E-13	1.2E-12	6.4E-11

Table 3 provides the plume characterization data for each release category.

The MACCS2 offsite dose and property damage risk quantification is executed for each release category source term. Consequences are calculated by MACCS2 for the first 24-hour period following onset of core damage. The code provides a complementary cumulative distribution function (CCDF) and the mean value results for each user-specified consequence. The CCDF presents the probability that a level of consequence is exceeded and can be used to evaluate the relative likelihood of a result compared to a mean or average value.

Table 3. Plume Characterization Data

Release Category (a)	Plume No.	Number of Plume Releases	Risk-Dominant Plume	Ref Time(b)	Plume Heat (W)	Plume Release Height (m)	Plume Duration (s)(c)	Plume Delay (s)(d)
RC1	1	2	1	0.0	0	0	3.6E+4	1.0E+5
RC1	2	2	1	0.5	0	0	8.6E+4	1.2E+5
RC2	1	2	1	0.0	0	0	5.3E+4	9.0E+3
RC2	2	2	1	0.5	0	0	8.6E+4	4.2E+4
RC3	1	2	1	0.0	0	0	4.4E+4	1.7E+5
RC3	2	2	1	0.0	0	0	8.6E+4	2.1E+5
RC4	1	2	1	0.0	0	0	3.2E+4	7.8E+4
RC4	2	2	1	0.5	0	0	8.6E+4	9.4E+4
RC5	1	2	1	0.0	0	0	6.0E+4	1.9E+5
RC5	2	2	1	0.5	0	0	8.6E+4	2.0E+5
RC6	1	2	1	0.0	0	0	7.3E+4	1.3E+3
RC6	2	2	1	0.5	0	0	8.6E+4	1.5E+4

In the SAMDA analysis, the mean values are used for the baseline risk profile. The population dose risk is calculated by multiplying the release category frequency by the mean value of the consequence result. Therefore, the overall population dose risk is the sum of the six release category risks and is reported in terms of person-sievert/year (person-Sv/y). Similarly, the offsite property damage risk is calculated based on the sum of the six individual mean property damage risks and reported in dollars/per reactor-year, ($/reactor-year). Radiation exposures are measured over a 50-mile radius from the plant site. The product of the radiation exposure and the monetary conversion factor of 2000 dollars/person-rem (equivalent to $3 million as the value of a statistical life, or VSL) is the monetary equivalent risk value (dollars/year). Note that one rem is 0.01 sievert. The MACCS2 results are summarized in Table 4, which shows variations for three different meteorological years from a severe accident at one of the Comanche Peak plants. Table 5 shows the breakdown of off-site consequences for the various release categories using 2006 meteorological data. Table 6 shows the off-site consequences breakdown for the various accident types, using 2006 meteorological data.

2.2 Identification of candidate SAMAs

The next step is to identify candidate SAMAs. Although the greatest level of risk reduction might be achieved by a major plant modification, lower cost alternatives might eliminate a substantial fraction of the risk and have a greater net benefit. In identifying SAMAs, the lowest cost means of achieving the functional objectives should not be overlooked. As an example, developing procedures to connect hydrogen igniters to portable on-site generators, rather than installing additional igniters with dedicated batteries, would be more cost-beneficial if it achieved the same reduction in risk. One key tool used in identifying SAMAs is the use of PRA importance measures, such as Risk Achievement Worth (RAW) to identify important basic events from the PRA (e.g., equipment failures and operator actions) and candidate SAMAs to address these basic events. In addition, a list of SAMAs that have been found to be cost-beneficial at other plants in the past should be reviewed to identify candidate SAMAs for the plant being analyzed.

Table 4. Off-Site Consequences from Severe Accidents at Comanche Peak 3 or 4

TABLE 7.2-5
SEVERE ACCIDENT ANALYSIS RESULTS SUMMARY WITHIN 50 MI OF CPNPP SITE(a)

Met Data Year	Dose Risk (person-rem/ RY)	Dollar Risk ($/RY)	Affected Land (hectares) (b)	Early Fatalities (per RY)	Latent Fatalities (per RY)	Water Ingestion Dose Risk (person-rem/ RY)
2001	2.21E-01	5.78E+02	2.66E-02	7.49E-08	1.85E-04	1.62E-02
2003	2.71E-01	6.62E+02	2.76E-02	7.43E-08	2.15E-04	1.52E-02
2006	3.00E-01	7.06E+02	2.70E-02	6.73E-08	2.39E-04	1.63E-02

a) All data are compiled from Tables 7.2-9, 7.2-10, and 7.2-11.

b) This value reflects the sum of affected land areas that have been multiplied by their release category frequency, whereas the affected land areas shown in the MACCS2 analysis are neither multiplied by release category frequency or summed. However, the same MACCS2 data were used as the basis for both values.

Table 5. Off-site consequences for the six release categories using 2006 meteorological data

TABLE 7.2-11
SEVERE ACCIDENT IMPACTS TO THE POPULATION AND LAND USING 2006 METEOROLOGICAL DATA

Release Category	Core Damage Frequency (per RY)	Dose-Risk (person-rem/RY)	Number of Early Fatalities (per RY)	Number of Latent Fatalities (per RY)	Affected Land Area (hectares)(a)	Cost-Risk (dollars/ RY)(b)	Water Ingestion Pathway (person-rem/ RY)
RC1	7.5E-09	2.93E-02	1.99E-09	1.97E-05	2.05E-03	9.90E+01	1.91E-03
RC2	2.1E-09	6.09E-03	2.46E-10	4.39E-06	7.01E-04	1.65E+01	1.27E-04
RC3	2.0E-08	8.96E-02	6.46E-08	1.27E-04	5.28E-03	3.38E+02	1.21E-02
RC4	1.1E-08	2.67E-02	4.70E-10	1.65E-05	2.44E-03	5.73E+01	6.90E-04
RC5	6.5E-08	1.48E-01	0.00E+00	7.09E-05	1.65E-02	1.95E+02	1.45E-03
RC6	1.1E-06	1.01E-03	0.00E+00	5.26E-07	7.69E-06	6.84E-03	2.41E-06
Total	1.2E-06	3.00E-01	6.73E-08	2.39E-04	2.70E-02	7.06E+02	1.63E-02

a) These values reflect affected land areas that have been multiplied by their release category frequency; whereas, the affected land areas shown in the MACCS2 analysis are not multiplied by release category frequency. However, the same MACCS2 data were used as the basis for both values.

b) The cost-risk accounts for the costs of evacuation, crops contaminated and condemned, milk contaminated and condemned, decontamination of property, and indirect costs resulting from the loss of use of property and incomes.

Table 6. Off-site consequences for accident types using 2006 meteorological data

TABLE 7.2-14
TOTAL SEVERE ACCIDENT HEALTH EFFECTS USING 2006 METEOROLOGICAL DATA(b)

Accident Type	Core Damage Frequency (per RY)(a)	Scaling Factor	Dose-Risk (person-rem/RY)	Number of Early Fatalities (per RY)	Number of Latent Fatalities (per RY)	Water Ingestion Pathway (person-rem/RY)
Internal Events	1.2E-6	1	3.00E-01	6.73E-08	2.39E-04	1.63E-02
Internal Fire	1.8E-6	1.50	4.50E-01	1.01E-07	3.59E-04	2.45E-02
Internal Flood	1.4E-6	1.17	3.51E-01	7.87E-08	2.80E-04	1.91E-02
LPSD	2.0E-7	0.167	5.01E-02	1.12E-08	3.99E-05	2.72E-03
Total	4.6E-6	-	1.15E-00	2.58E-07	9.17E-04	6.25E-02

a) Core damage frequency values are from Table 5 of the DC Applicant's Environmental Report (MHI 2007).

b) The values for internal fire, internal flood, and LPSD are calculated as described on page 7.2-7.

For the example plant being discussed here, a list of potential SAMDAs was compiled in the US-APWR Environmental Report [3], and utilized in Chapter 7 of the Comanche Peak Environmental Report [4], based on consideration of current pressurized water reactor (PWR) plant designs, information from the US-APWR PRA (proprietary), and design alternatives identified by design personnel. The resulting list contained 156 items that were subsequently analyzed to determine if there are cost-beneficial design alternatives that should be considered for the US-APWR design. The screening analysis identified 20 alternatives that are not applicable and 22 design alternatives that were already incorporated into the US-APWR design. Twenty-nine items were screened out because they were not design alternatives. Three items were not feasible because their cost would clearly outweigh any risk-benefit consideration. Another three items were similar in nature to other items and were combined with those items. Finally, there were 69 issues that were considered to have very low benefit due to their insignificant contribution to reducing risk. In summary, of the 156 total items analyzed, 10 items were not screened out using the previously mentioned screening criteria. The 10 SAMDAs that passed the screening process are as follows:

1. Provide additional dc battery capacity (at least one train of emergency dc power can be supplied for more than 24 hours.)
2. Provide an additional gas turbine generator (at least one train of emergency ac power can be supplied more than 24 hours.)
3. Install an additional, buried off-site power source
4. Provide an additional high pressure injection pump with independent diesel (with dedicated pump cooling)
5. Add a service water pump (add independent train)
6. Install an independent reactor coolant pump seal injection system, with dedicated diesel (with dedicated pump cooling)
7. Install an additional component cooling water pump (add independent train

8. Add a motor-driven feed water pump (with independent room cooling)
9. Install a filtered containment vent to remove decay heat
10. Install a redundant containment spray system (add independent train)

2.3 Estimation of risk reduction and implementation cost estimates

Once candidate SAMAs have been identified, an initial screening is performed to determine which ones may not be cost-beneficial. A rough implementation cost estimate is developed for each SAMA. If the cost exceeds the bounding condition of the MAB, then the SAMA is screened out from further consideration. In addition, candidate SAMAs from other plants that are not applicable to the plant being analyzed (e.g., due to design or risk-profile differences) are screened out.

For each remaining SAMA, a benefit assessment is performed to address how the change would affect relevant risk measures, including CDF, offsite population dose in person-Sv [person-rem], and offsite economic cost risk (OECR). This includes a description of how the change was implemented/credited in the PRA model (i.e., what changes were made to the basic events, fault trees, or event trees). For example, the impact of a procedural change might be estimated by reducing the associated human error probabilities. In some cases, bounding assumptions are used that capture the maximum possible benefit of the change, such as assuming that improvements to assure reactor cavity flooding would eliminate all containment failures due to core-concrete interactions.

A cost assessment is also performed for each SAMA. Cost estimates for hardware modifications can be taken from past studies performed for a similar plant, or developed on a plant-specific basis. Typically, screening estimates are used for initial assessments and refined as appropriate if a SAMA is potentially cost-beneficial. In general, hardware costs are several hundred thousand to a million dollars, and procedure changes range from ~$50K to $200K for complex changes with analysis and operator training impacts.

The licensee is expected to assess the impact of major uncertainties on the results to demonstrate the robustness of the conclusions. Sensitivity analyses are typically performed, examples of which include: (1) the estimated benefits are increased by the ratio of the 95th percentile CDF to mean CDF (to address uncertainty in the CDF analysis) and (2) alternative discount rates are used in the cost-benefit analysis (e.g., 7% versus 3%) to assess sensitivity of results to the assumed discount rate.

2.4 Estimation of maximum averted cost (maximum attainable benefit)

The net value of each SAMA is estimated, using NRC guidance in NUREG/BR-0058 [5] and NUREG/BR-0184 [6].

The net value of a particular SAMA can be generated from the following basic equation:

$$\text{Net Value} = \left(\text{APE} + \text{AOC} + \text{AOE} + \text{AOSC}\right) - \text{COE}$$

where:

APE = averted public exposure costs
AOC = averted offsite property damage costs
AOE = averted occupational exposure costs

AOSC = averted onsite costs = averted cleanup and decontamination costs (ACC) + averted replacement power costs (ARPC)

COE = cost of enhancement

This value is converted to present-day dollars by a factor (C) that discounts future losses to the present value as follows:

$$C = [1-e^{(-rt_f)}]/r$$

where:

C = present-value discount factor
r = real discount rate
t_f = years remaining until end of facility life (years)

The present-value discount factor is a multiplier applied to APE, AOC, AOE, and ACC. Following a severe accident, averted replacement power costs (ARPC) are considered for the remaining reactor lifetime. The single event costs are adjusted to account for all years of reactor service. A complicated expression is used to account for year-to-year variations.

Table 7 shows the US-APWR values of risk averted for all accident categories, and Table 8 shows the estimated costs for the various candidate SAMDAs for the US-APWR design. Table 8 also shows the computed value of the maximum averted cost, as well as the results of varying the discount rate (baseline is 7%) and the monetary equivalent of unit dose (baseline is $2000/person-rem). As can be seen, none of the SAMDAs are cost-beneficial.

Table 7. Value of Risk Averted for US-APWR Design

Table 1 – US-APWR Value of Risk Avoided

Cost Component	Internal Events	Internal Fire	Internal Flood	LPSD	Totals for All Events
Offsite Exposure Cost	$7.6k	$11.4k	$8.9k	$1.3k	$29.1k
Offsite Property Damage Cost	$0.1k	$0.2k	$0.1k	$0.02k	$0.5k
Onsite Exposure Cost	$0.6k	$0.9k	$0.7k	$0.1k	$2.3k
Cleanup and Decontamination Cost	$18.2k	$27.3k	$21.3k	$3.0k	$69.8k
Replacement Power Cost	$48.9k	$73.4k	$57.1k	$8.2k	$187.6k
Total (Maximum Averted Cost Benefit)	$75.5k	$113.2k	$88.1k	$12.6k	$289.3k

Table 8. SAMDA Benefit Calculation Results and Sensitivity Analysis

Table 12 – SAMDA Benefit Sensitivity Analyses

	Design Alternative	Cost Impact	Maximum Averted Benefit	Sensitivity of each SAMDA benefit			
				Baseline	Discount rate		Monetary equivalent of unit dose ($3000/person-rem)
					5%	3%	
1	Provide additional dc battery capacity	$2,000k	$289k	$116k	$188k	$304k	$124k
2	Provide an additional gas turbine generator	$10,000k		$116k	$188k	$304k	$124k
3	Install an additional, buried off-site power source	$10,000k		$118k	$193k	$312k	$127k
4	Provide an additional high pressure injection pump with independent diesel	$1,000k		$150k	$244k	$395k	$161k
5	Add a service water pump	$5,900k		$72k	$118k	$190k	$78k
6	Install an independent reactor coolant pump seal injection system, with dedicated diesel	$3,800k		$136k	$221k	$357k	$146k
7	Install an additional component cooling water pump	$1,500k		$72k	$118k	$190k	$78k
8	Add a motor-driven feed-water pump	$2,000k		$101k	$165k	$266k	$109k
9	Install a filtered containment vent to remove decay heat	$3,000k		$173k	$282k	$455k	$186k
10	Install a redundant containment spray system	$870k		$14k	$22k	$36k	$15k

Table 9 shows the total value of risk averted for Comanche Peak 3 or 4. Note that the totals are higher than for the design certification analysis because site-specific data were used to compute the off-site consequences and the replacement power costs. Nevertheless, none of the SAMDAs are cost beneficial when the discount rate is 7%. For a 3% discount rate, SAMDAs 4 and 10 would be needed to be considered further.

Table 9. Total Value of Risk Averted for Comanche Peak 3 or 4

Value	Internal Events	Internal Fire	Internal Flood	LPSD	Total
CDF (per RY)[(a)]	1.2E-06	1.8E-06	1.4E-06	2.0E-07	4.6E-06
CPNPP, 7% Discount Rate	$104,267	$156,401	$121,992	$17,413	$400,073
CPNPP, 3% Discount Rate	$274,852	$412,278	$321,577	$45,900	$1,054,607

a) Core damage frequency values are from Table 5 of the DC Applicant's Environmental Report (MHI 2007).

2.5 More detailed analysis for remaining SAMAs

The final step is a more detailed analysis of the potentially cost-beneficial SAMAs. A more detailed (i.e., more realistic and less bounding) evaluation is made of the potential benefits of the SAMA. For example, rather than assuming that the SAMA eliminates all CDF contributors, only those sequences relevant to the SAMA are included. A more detailed estimate of the cost of the proposed modification could be developed to include, for example, engineering support, training, hardware costs, and implementation costs. Additional guidance for conducting this step is available in a Nuclear Energy Institute (NEI) document NEI-05-01, Revision A [7]. The NRC staff has recommended that applicants for license renewal follow NEI-05-01 [8], Revision A, in the staff's Final License Renewal Interim Staff Guidance LR-ISG-2006-03 [9]. Note that the example SAMDA analysis discussed in this paper did not take this additional step because none of the SAMDAs were determined to be cost-beneficial (with a 7% discount rate) during the initial stage.

3. INSIGHTS FROM SAMA AND SAMDA EVALUATIONS

The NRC has realized many insights from reviewing and evaluating the various SAMA and SAMDA analyses submitted by the nuclear industry. These are summarized below.

3.1 Insights from SAMA evaluations for license renewal

The PRAs for the currently-operating plants all report low values for CDF, as can be seen in Table 10. The CDF is divided among the various release categories for the MACCS2 calculations. Typically, the dominant release category is one where the containment does not fail and is not bypassed, and the off-site consequences are very low for this release category. See, for example, Table 1 above, where more than 90% of the CDF is in RC6 and fission product release is at the design leakage rate. Consequently, the release fraction is very low and the overall contribution to risk is very small. The largest contribution to risk is from RC3, because the source term is high (for example, 47% cesium release fraction during the first plume, which lasts from 48 hours to 72 hours after the start of the accident). The CDF for this release category is very low, however, because it is highly likely that containment heat removal would be restored by 48 hours.

Table 10. Ranges of results from SAMA analyses for license renewal

	Average	Ranges
CDF (/yr)	4.0×10^{-5}	$1.9 \times 10^{-6} – 3.3 \times 10^{-4}$
Population Dose (person-Sv/year)	0.15	0.006 – 0.69
$/event	$2.8 billion	$49 million – $12 billion
$/person-Sv	$220,000	$69,000 - $670,000
Total MAB	$1.7 million	$110,000 – $8.7 million

It has also been observed that the utilities have addressed past known weaknesses by making design improvements and put in place effective procedures and training programs, that SAMAs typically only act on one contributor, while risk is generally driven by multiple contributors, and that implementation costs are high for design retrofits. Therefore, it is difficult to identify additional changes that substantially reduce risk and are cost-beneficial. In practice, cost-beneficial changes usually limited to procedural changes and limited hardware changes. Finally, averted onsite costs are important, and promote preventative SAMAs.

3.2 Insights from new reactor SAMDA evaluations

The most important insight is that the new designs include severe accident prevention and mitigation features not found in first-generation plants. These address issues raised by the Commission that must be resolved in the design. Therefore, the overall severe accident risk is already significantly lowered and it is difficult to identify additional cost-effective design features. However, some potential exists for impacts arising from adopting design departures and from site-specific external events.

4. REFERENCES

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